NON-PUBLIC?: N
ACCESSION #: 8905240488
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Catawba Nuclear Station, Unit 1 PAGE: 1 of 7

DOCKET NUMBER: 05000413

TITLE: Reactor Trip and Safety Injection During Auxiliary Safeguards Testing
Due to Inappropriate Action and Failure of a Power Operated Relief
Isolation Valve
EVENT DATE: 03/05/89 LER #: 89-008-01 REPORT DATE: 05/09/89

OPERATING MODE: 1 POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION
50.73(a)(2)(iv) and OTHER - 50.72(b)(2)(ii)

LICENSEE CONTACT FOR THIS LER:
NAME: R. L. White, Chairman, Catawba Safety TELEPHONE: (803) 831-3393
Review Group

COMPONENT FAILURE DESCRIPTION:
CAUSE: X SYSTEM: SO COMPONENT: XI MANUFACTURER: R335
X SO XCV R200
A SO V D232

REPORTABLE TO NPRDS: N
Y
N
SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On March 5, 1989, at 2140 hours, a Unit 1 Reactor Trip occurred after exceeding the Over Power Delta Temperature setpoint. In preparation for Auxiliary Safeguards Testing of Steam Line Isolation Valves, the Operator at the Controls mistakenly depressed the CLOSE pushbutton for the Main Steam Loop A Isolation Valve while attempting to close the Steam Generator 1A Power Operated Relief Isolation Valve. The two pushbuttons are adjacently located. The Power Operated Relief Isolation Valve had failed to close on two previous pushbutton actuations during this test. The Main Steam Isolation Valve was immediately reopened, but as other relief valves had also opened, Safety Injection subsequently actuated on low rate compensated steam pressure. Unit response to the trip and Safety Injection was as designed. Control Room

personnel entered the appropriate Emergency Procedures to stabilize the Unit and recover from the Safety Injection. The application of protective covers to certain control devices has been reviewed. The Power Operated Relief Isolation Valve actuator was found to have a worn stem nut which was subsequently replaced. The Unit was operating in Mode 1, Power Operation, at 100% power at the time of the trip.

END OF ABSTRACT

BACKGROUND

PT/1/A/4200/09A, Auxiliary Safeguards Test Cabinet Periodic Test, is performed quarterly to verify operability of the final actuating device for nuclear safety related equipment while the Unit is in service without disturbing operation of the Unit. Section 12.3 verifies proper response to a Steam Line Isolation generated by Solid State Protection System EIIS:JE! Train A Slave Relay EIIS:RLY! K616 actuation. In this section, closure of the Steam Generator Power Operated Relief Valves EIIS:V! (PORV) is verified. The Steam Generator Power Operated Relief Isolation Valves are initially closed to allow the PORVs to be opened for testing.

Safety Injection may be initiated by either low steam line pressure (725 psig) when Reactor Coolant (EIIS:AB) pressure is greater than 1955 psig, or from high steam pressure rate (100 psi/sec) when Reactor Coolant pressure is less than 1955 psig.

Steam line pressure is also rate compensated, so that a rapid drop in pressure may initiate Safety Injection at an actual steam pressure of well above 725 psig.

EVENT DESCRIPTION

On March 5, 1989, Unit 1 was operating at 100% power. At approximately 2130 hours, Performance and Operations were preparing to conduct Section 12.3 of PT/1/A/4200/09A, Auxiliary Safeguards Test Cabinet Periodic Test. This section verifies that Steam Generator (S/G) PORVs will cycle closed from the open position when slave relay K616 Train A is actuated. To enable this test to be performed while the Unit is operating, the PORV Isolation Valves are initially closed, allowing the PORVs to be opened. As directed by the PT, the Operator at the Controls (OATC) depressed the CLOSE pushbutton for 1SV25B, 26B, 27A, and 28A, S/G 1D, 1C, 1A, and 1B PORV Isolation Valves at 2139 hours. No response was noted from 1SV27A. The OATC again depressed the 1SV27A pushbutton EIIS:XIS!, and the valve did not respond. While attempting to actuate the pushbutton again, the OATC inadvertently depressed an adjacent

CLOSE pushbutton for 1SM7, Main Steam EIIS:SB! Loop A Isolation Valve, at 2140:32 hours. The inappropriate action was immediately recognized and the OATC reopened 1SM7. 1SM7 began opening at 2140:44 hours. 1SV20 and 1SV21, S/G 1A Code Safety Relief Valves, opened as steam pressure increased to approximately 1185 psig.

At 2140:44:546 hours, the Reactor tripped due to exceeding the Over Power Delta Temperature Loop 2 setpoint. The Condenser Steam Dump EIIS:SO! Valves immediately began cycling to relieve the steam pressure increase, and the Turbine EIIS:TRB! automatically tripped. Steam pressure began to decrease rapidly as steam was being released through Condenser Steam Dump Valves and Code Safety Relief Valves on S/Gs 1A and 1B.

At 2140:45:775 hours, Steam Line Loop A Low Pressure Safety Injection actuated as the rate compensated steam pressure decreased to the 725 psig setpoint. Actual S/G 1A pressure was approximately 1110 psig. Both Diesel Generator (D/G) Load Sequencers EIIS:EB! actuated on accelerated sequence, and the D/Gs EIIS:ENG! started as expected. Feedwater Isolation, Main Steam Isolation, and Phase A Containment Isolation actuated and the appropriate valves closed. Both Main Feedwater EIIS:SJ! Pumps EIIS:P! tripped as expected. In response to the Safety Injection signal, both Annulus Ventilation EIIS:VD! Fans EIIS:BLO! started, Centrifugal Charging Pump 1A started, both Safety Injection EIIS:BQ! Pumps started, both Residual Heat Removal EIIS:BP! Pumps started, Nuclear Service Water EIIS:BI! Pumps 1B, 2A, and 2B started, Component Cooling Water EIIS:CC! Pumps 1A1 and lA2 started, and both Motor Driven Auxiliary Feedwater EIIS:BA! Pumps started as expected. Charging Pump suction automatically aligned to the borated Fueling Water Storage Tank EIIS:ACC!. Emergency Procedure EP/1/A/5000/01, Reactor Trip or Safety Injection, was entered, and an Unusual Event was declared at 2143 hours.

EP/1/A/5000/01B, S/I Termination Following Spurious S/I, was entered after EP/1/A/5000/01 was completed. At 2147 hours, Control Room personnel reset Safety Injection, the Load Sequencers, and Phase A Containment Isolation signal. Both Residual Heat 'Removal Pumps were secured, both Safety Injection Pumps were secured, Centrifugal Charging Pump 1A was secured, and the Charging Pump Cold Leg Injection Isolation Valves were closed as directed by the EP.

As the S/G PORV Isolation Valves had been closed prior to the trip to support testing, Reactor Coolant System temperature was maintained at approximately 565 degrees F by opening of S/G Code Safety Relief Valves. At 2151 hours, the S/G PORV Isolation Valves were opened by Control Room personnel, and the S/G PORVs were used to cooldown the Reactor Coolant System to approximately 557 degrees F. At 2154 hours, Charging Suction was realigned to the normal source. At 2205 hours, the Unusual Event was terminated. Nuclear Service

Water Pump 1B and Component Cooling Pump 1A2 were secured at approximately 2224 hours.

On March 7, 1989, at 0045 hours, Main Feedwater Pump 1A was returned to service and the Auxiliary Feedwater System was returned to Standby Readiness.

The Unit entered Mode 1, Power Operation, on March 8, 1989, at 0109 hours.

CONCLUSION

The cause of this event is attributed to an inappropriate action due to lack of attention to detail. The OATC mistakenly actuated the CLOSE pushbutton for 1SM7 while attempting to actuate the control for 1SV27. The two pushbuttons are adjacently located on the main control board, and both valves were indicating OPEN.

A contributing cause of the event was the failure of 1SV27A's actuator EIIS:XCV!. Work Request 50079 OPS was issued and completed to

investigate/repair 1SV27. The actuator was found to cycle but the valve did not. The stem nut connecting the actuator to the valve was found to be stripped due to wear, subsequently replaced, and the actuator was functionally verified prior to Unit restart. 1SV27A is a 6 inch gate valve, manufactured by Rockwell International, figure 1711 (WCC) FJMNPQTY. The actuator is a Limitorque SMB-00-15. No failures of stem bushings or drive nuts have been reported to NPRDS.

Work Request 6914 PRF was issued and completed to investigate/repair position indication for 1SV20. A broken sensing line for downstream pressure switch 1SVPS5150 was found. The instrument isolation valve was also replaced. The sensing line appeared to have been stepped on.

1SV13, S/G 1B PORV, would not open during the incident. Work Request 50080 OPS was issued to investigate/repair 1SV13. The valve was stroked while isolated under pressure, after which it was unisolated and partially stroked. Both valve strokes were successful. Problem Investigation Report 1-C89-0111 was then written to investigate the behavior of 1SV13 during the transient. Nuclear Station Modification 50395 was written to increase the pilot area of all of the S/G PORVs, and this modification has been performed on 1SVI3. In addition, Exempt Change CE-2253 was initiated to install instrument taps to enable the measurement of differential pressure across the valve. This modification has also been performed.

1SV14, S/G 1B Code Safety Relief, lifted five times during the transient. The first actuation of the valve occurred as expected with steam header pressure reaching 1180 psig: The liftpoint is 1175 psig (+/- 1%). The subsequent lifts occurred at approximately 1140 psig. Problem Investigation Report 1-C89-0106 was initiated to identify the questionable response of the valve. Evaluation of the PIR concluded that heating of the valve internals may decrease the actual liftpoint by up to 3%, and that the valve's performance was acceptable for its intended safety function.

Operations has reviewed main control board pushbutton switches which will initiate a non-recoverable transient and provided protective covers to minimize physical misactuation. The covers were added to switches generally associated with Reactor Coolant Pumps, Main Feedwater, and Main Steam Isolation controls.

In the past twelve months, there have been several ESF actuations due to lack of attention to detail (see LERs 413/88-18, 413/89-01, and 414/90-04), so this is considered to be a recurring event. ESF actuation due to valve motor operator failure has not occurred in the previous twelve months, and is not considered to be a recurring event.

A review of previous incidents during the past three years showed that Safety Injections (S/I) are a recurring problem at Catawba. LER 414/88-003 describes an S/I caused by not following Station Directives regarding tagouts. LER 414/86-049 describes a spurious S/I caused by dirty contacts in the Westinghouse

7300 Process Control Cabinet lead/lag card (EIIS:IMOD). LER 414/86-041 describes an S/I due to the malfunction of a Main Steam Bypass to Condenser (SB) valves. LER 414/86-028 describes an S/I caused by S/G PORV controls and procedural problems. LER 414/89-01 described an S/I initiated due to an inappropriately placed jumper occurring during troubleshooting. LER 414/89-04 described an S/I initiated during Unit cooldown when inadequate attention was directed to the steam pressure decrease, and no alarm was designed to alert the Operator that rate compensated steam pressure indication was nearing the S/I actuation setpoint.

CORRECTIVE ACTION

IMMEDIATE

The OATC immediately reopened 1SM7, but was unsuccessful in preventing the Reactor Trip.

SUBSEQUENT

1) Control Room personnel stabilized the Unit and recovered from the S/I in accordance with Emergency Procedures.

2) 1SV27A actuator stem nut was replaced.

3) Supervision reviewed the event with the OATC.

4) Protective covers were added to certain main control board devices to minimize the potential for misactuation.

PLANNED

1) The responsibility for resolution of all inadequate post-trip responses will be assigned and all such items will be identified on the Station Commitment Index.

SAFETY ANALYSIS

Upon inadvertent closure of S/G 1A Main Steam Isolation Valve (MSIV), 1SM7, steam pressure immediately increased. S/Gs 1B, 1C, and 1D steam pressure began to decrease since turbine load remained the same. Also, since feedwater flow to S/G 1A significantly decreased (due to the rise in steam pressure), increased feedwater flow was forced to S/Gs 1B, 1C, and 1D. Steam flow from S/G 1A was temporarily terminated, and steam flow from S/Gs 1B, 1C, and 1D increased. Due to the increased feedwater flow to and steam flow from S/Gs 1B, 1C, and 1D, Reactor Coolant (NC) System Loop A Tcold increased, and NC Loops B, C, and D Tcold decreased. Therefore, NC Loops B, C, and D delta-T (Thot-Tcold) increased. Also, after closure of 1SM7, the Operator reopened the valve, likely furthering the decrease in NC loops B, C, and D Tcold (and consequent increase in delta
T). NC Loop D delta-T reached the Overpower
Delta-T trip setpoint and initiated a Reactor Trip from 100% full power. Upon the initial increase in S/G

1A steam pressure during closure of valve 1SM7, S/G 1A Code Safety Relief Valves, 1SV20 and 1SV21 opened. Also upon steam pressure increase, the steam dump to condenser valves opened to dump steam, with the exception of 2 valves which malfunctioned (and one steam dump to condenser valve had been manually isolated), The S/G 1C and 1D PORVs also automatically opened, but did not dump steam because they had been previously manually isolated for testing. The S/G 1A PORV opened to dump steam. The S/G 1B PORV (lSV13) would not open during the transient. 1SV13 had also been manually isolated for testing. During the consequent steam pressure decrease due to the code safety relief and steam dump to condenser valves opening, a Safety Injection signal was initiated upon steam line A pressure rate of decrease signal within approximately 1 second after the Reactor Trip signal.

Upon Reactor Trip signal, all of the control rods fell to the bottom of the core, reducing power to decay heat level. The Reactor Trip Breakers opened within 71 milliseconds of the Reactor Trip signal. Upon Safety Injection, the following major interlocks and equipment were actuated:

* Diesel Generator automatic start and sequencer actuation (accelerated sequence loading)

* Containment Isolation

* Main Steam Isolation

* Safety Injection Pumps 1A and 1B

* Residual Heat Removal Pumps 1A and 1B

* Centrifugal Charging Pump 1A (Pump 1B previously operating)

* Component Cooling Water-Pumps 1A1 and 1A2

* Nuclear Service Water Pumps 1B, 2A, and 2B (Pump 1A previously operating)

* Annulus Ventilation System

* Auxiliary Building Ventilation System Safety Injection alignment

* Motor Driven Auxiliary Feedwater Pumps 1A and 1B

Upon closure of 1SM7, NC Loop A Tave increased 10 degrees F to a maximum value of 598 degrees F, and NC Loops B, C, and D Tave decreased approximately 3 degrees F. Upon Reactor Trip, average NC Tave decreased to approximately 566 degrees F, and fluctuated due to S/G Code Safety Relief Valve and S/G PORV cycling. NC Tave stabilized at 555 degrees F within 30 minutes post-trip, 2

degrees F from the no-load target of 557 degrees F. Upon closure of 1SM7, pressurizer pressure increased 10 psi to a maximum value of 2240 psig, and decreased to a minimum value of 2090 psig upon Reactor Trip. Pressurizer pressure stabilized at 2200 psig within 30 minutes post-trip, 35 psi from the no-load target of 2235 psig. Upon Reactor Trip, pressurizer level decreased to a minimum value of 39%, and subsequently increased to a maximum value of 73% due to Safety Injection. Pressurizer level stabilized at 53% within 30 minutes post-trip, 28% from the no-load target of 25%. S/G 1A steam pressure immediately increased to a maximum value of 1185 psig upon closure of 1SM7, while S/Gs 1B, 1C, and 1D steam pressure decreased to an average minimum value of approximately 950 psig, and subsequently increased to an average value of approximately 1165 psig. Steam pressure fluctuated due to Code Safety Relief Valve and PORV cycling and within 30 minutes post-trip had stabilized at 1040 psig, 50 psig from the no-load target of 1090 psig. S/G narrow range levels remained on-scale throughout this event.

The Safety Injection signal initiated automatic Main Feedwater pump trip and Motor Driven Auxiliary Feedwater pump autostart. Steam dump and heat removal was accomplished by unisolating the S/G PORVs and manually cycling S/G 1C and 1D PORVs. The Reactor Coolant was 33 degrees F subcooled at the point of minimum Reactor Coolant System pressure. Valves 1NV252A and 1NV25AB automatically opened upon Safety Injection to swap Centrifugal Charging Pump suction from the Volume Control Tank to the Fueling Water Storage Tank. Valves 1NI9A and 1NI10B automatically opened upon Safety Injection to allow Centrifugal Charging Pumps to discharge 2000 ppm Borated water directly to the cold legs, and NC System coolant inventory was maintained. Adequate heat sink for core decay heat removal was available and maintained at all times.

The Catawba FSAR, Section 15.2.4, describes inadvertent closure of Main Steam Isolation Valves, stating that an inadvertent closure of an MSIV would result in a Turbine trip. An MSIV closure and the ensuing transient is therefore bounded by the Turbine Trip scenario as discussed in the Catawba FSAR, Section 15.2.3.

S/G 1B Code Safety Relief Valve, 1SV14, initially opened upon steam pressure increase at its design pressure of 1175 psig + 1%. Steam flow through this valve heated the valve internals, thereby reducing the lift setpoint by approximately 3% during subsequent lifts. All subsequent lifts of this valve occurred at 1140 to 1146 psig, showing a constant setpoint after the initial decrease. This reduction in lift setpoint was in a conservative direction; therefore, there are no overpressure concerns with respect to this valve.

All plant safety equipment was available throughout this transient. The value of the usage factor for the cold leg injection nozzle does not exceed 0.70. Catawba Unit 1 has experienced four previous Safety Injections. Therefore, this event constitutes the fifth actuation cycle to date. The cooldown limits of 100 degrees F per hour for the Reactor Coolant System and 200 degrees F per hour for the pressurizer were not exceeded. Integrity of the fuel cladding, Reactor Coolant System, and Containment structure was maintained at all times. The health and safety of the public were not affected by this event.

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NUCLEAR PRODUCTION

May 9, 1989

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: Catawba Nuclear Station, Unit 1
Docket No. 50-413
LER 413/89-08, Revision 1

Gentlemen:

Pursuant to 10 CFR 50.73 Section (a) (1) and (d), attached is Revision 1 to Licensee Event Report 413/89-08 concerning a Reactor trip and safety injection during auxiliary safeguards testing. This event was attributed to an inappropriate action and the failure of a Power Operated Relief Valve.

This event was considered to be of no significance with respect to the health and safety of the public.

Very truly yours,

Hal B. Tucker

JGT/3/U1-89-8R

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